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Constellation Energy
R.E. Ginna Nuclear Power Plant

April 14, 2005

Ms. Donna M. Skay
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Subject: LER 2005-001, Failure of ADFCS Power Supplies Results in Plant Trip
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Ms. Skay:

The attached Licensee Event Report (LER) 2005-001 is submitted in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(iv)(A), which requires a report of "Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section".

This event has in no way affected the public's health and safety.

Very truly yours,

Mary G. Korsnick

Mary G. Korsnick

IE22

1001290

xc: Ms. Donna M. Skay (Mail Stop O-8-C2)
Project Directorate I
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U.S. NRC Ginna Senior Resident Inspector

NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Failure of ADFCS Power Supplies Results in Plant Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
02	16	2005	2005	- 001 -	00	04	14	2005	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Thomas L. Harding, Senior Licensing Engineer	(585) 771-3384

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	JB	RJX	ACDC	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On February 16, 2005, at approximately 2112 EST, with the plant in Mode 1 at approximately 100% steady state reactor power, the reactor automatically tripped. The Control Room operators performed the appropriate actions of procedures E-0 and ES-0.1. Following the reactor trip, all safety systems operated as designed, with the exception of the control function for the main steam atmospheric relief valves (ARVs). The reactor was stabilized in Mode 3.

The tripping of the reactor was caused by a turbine trip as the result of an anticipated-transient-without-scram (ATWS) mitigation actuation circuitry (AMSAC) signal. The AMSAC signal was the result of low feedwater flow signals, caused by the failure of redundant advanced digital feedwater control system (ADFCS) power supplies. The ARV automatic and remote manual operation was also affected by the failed power supplies.

Corrective action to prevent recurrence is outlined in Section V.B.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PRE-EVENT PLANT CONDITIONS:

On February 16, 2005, the R.E. Ginna Nuclear Power Plant (Ginna) was in Mode 1 at approximately 100% steady state reactor power. An investigation of an Advanced Digital Feedwater Control System (ADFCS) power supply failure alarm was in progress.

II. DESCRIPTION OF EVENT:

A. EVENT (INCLUDING DATES AND APPROXIMATE TIMES OF OCCURRENCES):

On February 16, 2005 at 2112 EST with the plant in Mode 1 operating at approximately 100% steady state reactor power, the primary power supply for ADFCS failed. Since there was no redundant power supply available at this time due to the earlier failure of the ADFCS secondary power supply, the primary power supply failure caused both main feedwater regulating valves (MFRVs) to go closed. The power supply failure also resulted in a low feedwater flow input to all four of the anticipated-transient-without-scram (ATWS) mitigation actuation circuitry (AMSAC) flow signals, which satisfied the 3 of 4 low flow inputs for starting the AMSAC timer. After the AMSAC time delay for full power operation (~25 seconds) was completed, AMSAC initiated a turbine trip. The turbine trip from AMSAC initiated a reactor trip signal which caused the reactor to trip.

The Control Room operators performed the appropriate actions of Emergency Operating Procedure E-0 (Reactor Trip or Safety Injection). The operators then transitioned to Emergency Operating Procedure ES-0.1 (Reactor Trip Response) when it was verified that both reactor trip breakers were open, all control and shutdown rods were inserted, and safety injection was not actuated or required.

B. INOPERABLE STRUCTURES, COMPONENTS, OR SYSTEMS THAT CONTRIBUTED TO THE EVENT:

ADFCS Secondary Power Supply

C. OTHER SYSTEMS OR SECONDARY FUNCTIONS AFFECTED:

Main Steam Atmospheric Relief Valves (ARVs)

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. METHOD OF DISCOVERY:

The reactor trip was immediately apparent due to plant response, alarms, and indications in the Control Room.

E. SAFETY SYSTEM RESPONSES:

Both motor driven auxiliary feedwater (AFW) pumps and the turbine driven AFW pump started as per design, due to a starting signal from AMSAC, and functioned properly. Due to the ADFCS failure, the automatic and remote manual control of the Main Steam ARVs was not available. Local manual operation of the ARVs remained available and met the requirements for their safety function.

III. CAUSE OF EVENT:

The immediate cause of the reactor trip was a turbine trip caused by an AMSAC signal. The AMSAC signal was the result of redundant ADFCS power supply failures.

The underlying cause of the redundant ADFCS power supply failures has been determined to be age related degradation, which led to a loss of cooling fans and subsequent power supply failure. Post trip testing of the power supplies showed that both power supply fans had degraded and/or failed which resulted in fatal power supply internal temperatures and subsequent power supply failures within a short time period. The existing preventative maintenance program, which verified power supply output voltages within specification, did not fully address age/cooling issues.

This event is NUREG-1022 Cause Code (E), "Management/Quality Assurance deficiency".

IV. ASSESSMENT OF THE SAFETY CONSEQUENCES OF THE EVENT:

This event is reportable in accordance with 10 CFR 50.73, Licensee Event Report System, item (a)(2)(iv)(A), which requires a report of, "Any event or condition that resulted in a manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section". The automatic reactor trip is an actuation of the reactor protection system, and AFW pump starts are actuations of a PWR auxiliary feedwater system.

An assessment was performed considering both the safety consequences and implications of this event with the following results and conclusions:

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

There were no operational or safety consequences or implications attributed to the reactor trip because:

o The two reactor trip breakers opened as required.

o All control and shutdown rods inserted as designed.

o The plant was stabilized in Mode 3.

o The closure of both MFRVs at full power represents a decrease in secondary side heat removal. Therefore, the plant transient was reviewed against the following Ginna Updated Final Safety Analysis Report (UFSAR) design basis Loss of Secondary Side Heat Removal transients:

- Loss of External Electrical Load
- Loss of all Alternating Current Power to Station Auxiliaries
- Loss of Normal Feedwater Flow Transient

These transients represent overheating events for the Reactor Coolant System (RCS) where reactor trip is provided by the Reactor Protection System (RPS). These UFSAR transients were examined and compared to the plant response for the actual event. The plant behavior was found to be consistent with, and bounded by, the events detailed in the accident analysis. All of the UFSAR transients were found to be bounding due to a combination of less limiting actual plant conditions and proper operation of plant equipment in responding to the plant trip. Specifically, the reactor trip occurred directly as a result of the turbine trip which mitigated the RCS overheating transient. Additionally, the AMSAC actuation caused an early initiation of AFW flow to both steam generators (SGs).

o The Ginna Improved Technical Specifications (ITS) Limiting Conditions for Operation (LCOs) and Surveillance Requirements (SRs) were reviewed with respect to the post trip review data. The following are the results of that review:

(a) Pressurizer (PRZR) pressure decreased below 2205 PSIG during the transient after the reactor trip. During this time the plant was in Mode 3, where ITS LCO 3.4.1 is no longer applicable. Therefore, compliance with ITS was maintained. Minimum PRZR pressure was approximately 2115 PSIG, and PRZR pressure was restored > 2205 PSIG within approximately nine (9) minutes.

(b) After the reactor trip, the RCS cooled down to approximately 540 degrees F and was subsequently stabilized at 547 degrees F. The cooldown was within the limits

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

of ITS LCO 3.4.3. In addition, the required shutdown margin was maintained at all times during the RCS cooldown.

(c) After the reactor trip, levels in both the "A" and "B" SGs decreased below 16%. During this time the plant was in Mode 3, where compliance with ITS LCO 3.4.5 is required. An operable RCS loop consists of an operable reactor coolant pump and an operable SG, which has the minimum water level of 16%. During the time when the level in a SG was < 16%, the SG was considered inoperable. For approximately sixty two (62) minutes, the plant was in ITS CONDITION 3.4.5.C, with both RCS loops inoperable. The REQUIRED ACTION for CONDITION 3.4.5.C was performed, and SG levels in both SGs were restored above 16% by approximately 2236.

o The potential for a loss of the automatic and remote manual control of the Main Steam ARVs was assumed during the design of the ADFCS system. In the event of a failure of the automatic and remote manual controls to control steam generator pressure, backup solenoid valves energize to open the Main Steam ARVs at 1060 psig. When the pressure decreases to 1005 psig, the backup solenoid valves de-energize causing the valves to close. Local manual operation of the Main Steam ARVs remained available and met the requirements for their safety function.

Based on the above and the review of post trip data and past plant transients, it can be concluded that the plant operated as designed, that there were no unreviewed safety questions, and that the public's health and safety was assured at all times.

V. CORRECTIVE ACTIONS:

A. ACTION TAKEN TO RETURN AFFECTED SYSTEMS TO PRE-EVENT NORMAL STATUS:

o The Control Room operators performed the appropriate actions of Emergency Operating Procedures E-0 and ES-0.1, and the plant was stabilized in Mode 3.

o Both failed ADFCS power supplies were removed and replaced with spare power supplies. The power supplies were re-energized, output voltages and ripple were verified to be within operating specifications, and ADFCS was returned to normal operating status.

o In addition, the ADCFS Rack cooling fan array was also replaced as a conservative measure. One fan of the nine fan array removed from the cabinet had failed.

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

B. ACTION TAKEN OR PLANNED TO PREVENT RECURRENCE:

NOTE: There are no NRC regulatory commitments in this Licensee Event Report.

- o Replace all ADFCS power supplies and fan arrays on a four refueling outage frequency.
- o Routinely monitor power supply output voltage and ripple against design specifications.
- o Purchase and/or refurbish spare power supplies.
- o Replace remaining original vintage ADFCS power supplies.
- o Evaluate the development of a new power supply design with a split chassis for better maintainability.
- o Review other power supplies in plant equipment that may rely upon muffin fans for cooling and evaluate replacement per a preventative maintenance program.

VI. ADDITIONAL INFORMATION:

A. FAILED COMPONENTS:

The failed ADFCS power supplies are nominally rated as 12V @ 70A. They were manufactured by ACDC Electronics Inc. and carry the "JF series" model designation. Model No. JF751B - 3000-0000.

B. PREVIOUS LERs ON SIMILAR EVENTS:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events with the same root cause at Ginna Station could be identified.